**AmerGen**

An Exelon/British Energy Company

**Clinton Power Station**

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10CFR50.73

U-603462

June 26, 2002

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Attn: Document Control Desk

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Clinton Power Station
Licensee Event Report No. 2000-007-01

Enclosed is Licensee Event Report (LER) No. 2000-007-01: Unknown Division 1 Logic Circuit Card Failure Goes Undetected During Division 2 Surveillance Test Due to Inadequate Indication/Annunciation and Completes Actuation Logic Resulting in Main Steam Line Containment Isolation and Reactor Scram. The enclosed report has been revised to change the cause of event and corrective actions. This report is being submitted in accordance with the requirements of 10CFR50.73.

Respectfully,

M. J. Pacilio
Site Vice President
Clinton Power Station

RSF/blf

Enclosure

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector - Clinton Power Station
Illinois Department of Nuclear Safety – Office of Nuclear Facility Safety

IE22

NRC FORM 366 (7-2001)

U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004

Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE
Unknown Division 1 Logic Circuit Card Failure Goes Undetected During Division 2 Surveillance Test Due to Inadequate Indication/Annunciation and Completes Actuation Logic Resulting in Main Steam Line Containment Isolation and Reactor Scram

| 5. EVENT DATE | | | 6. LER NUMBER | | | 7. REPORT DATE | | | 8. OTHER FACILITIES INVOLVED | |
|---|---|---|---|---|---|---|---|---|---|---|
| MO | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REV NO | MO | DAY | YEAR | FACILITY NAME | DOCKET NUMBER |
| | | | | | | | | | None | 05000 |
| 12 | 18 | 2000 | 2000 | – 007 – | 01 | 06 | 26 | 02 | None | 05000 |

| 9. OPERATING MODE | 1 |
|---|---|
| 10. POWER LEVEL | 100 |

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

| | | | | | | |
|---|---|---|---|---|---|---|
| | 20.2201(b) | | 20.2203(a)(3)(ii) | | 50.73(a)(2)(ii)(B) | 50.73(a)(2)(ix)(A) |
| | 20.2201(d) | | 20.2203(a)(4) | | 50.73(a)(2)(iii) | 50.73(a)(2)(x) |
| | 20.2203(a)(1) | | 50.36(c)(1)(i)(A) | X | 50.73(a)(2)(iv)(A) | 73.71(a)(4) |
| | 20.2203(a)(2)(i) | | 50.36(c)(1)(ii)(A) | | 50.73(a)(2)(v)(A) | 73.71(a)(5) |
| | 20.2203(a)(2)(ii) | | 50.36(c)(2) | | 50.73(a)(2)(v)(B) | OTHER Specify in Abstract below or in NRC Form 366A |
| | 20.2203(a)(2)(iii) | | 50.46(a)(3)(ii) | | 50.73(a)(2)(v)(C) | |
| | 20.2203(a)(2)(iv) | | 50.73(a)(2)(i)(A) | | 50.73(a)(2)(v)(D) | |
| | 20.2203(a)(2)(v) | | 50.73(a)(2)(i)(B) | | 50.73(a)(2)(vii) | |
| | 20.2203(a)(2)(vi) | | 50.73(a)(2)(i)(C) | | 50.73(a)(2)(viii)(A) | |
| | 20.2203(a)(3)(i) | | 50.73(a)(2)(ii)(A) | | 50.73(a)(2)(viii)(B) | |

12. LICENSEE CONTACT FOR THIS LER

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| J. C. Wemlinger, Corrective Action Coordinator Lead | (217) 937-3846 |

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

| CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| X | AD | RLY | F180 | Y | X | SJ | IMOD | G080 | Y |

| 14. SUPPLEMENTAL REPORT EXPECTED | | | | 15. EXPECTED SUBMISSION DATE | MONTH | DAY | YEAR |
|---|---|---|---|---|---|---|---|
| | YES (If yes, complete EXPECTED SUBMISSION DATE) | X | NO | | | | |

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

A Containment Main Steam Line isolation (Group 1) and subsequent reactor scram from 100 percent reactor power occurred during a Division (Div) 2 Main Steam Line tunnel ambient temperature channel functional test. The logic for the isolation was completed when a Div 2 channel functional test signal combined with a pre-existing Div 1 trip signal. The Div 1 trip signal was a result of a circuit card failure and was unknown at the time of the test. The isolation initiated a Reactor Scram. During recovery from the event, an additional scram occurred on low reactor water level. While reestablishing the main condenser as a heat sink, a deficient procedure lacked provisions for opening both the inboard and outboard main steam isolation valves and an at risk revision of the procedure was initiated. During the revision process, poor command and control allowed reactor pressure to increase and water level to decrease to the scram setpoint. The cause of the Group 1 and scram was inadequate indication / annunciation for the technician to detect a pre-existing fault in an alternate channel. The cause of the additional scram was poor command and control. Corrective action for this event includes changing circuit design, revising procedures, counseling the main control room crew, providing remedial training, and developing a comprehensive corrective action plan to address the operator performance issues.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

DESCRIPTION OF EVENT

On December 18, 2000, the plant was in Mode 1 (Power Operation) at 100 percent reactor power. Instrumentation and Control technicians were performing surveillance procedure CPS 9532.07, "MSL Ambient Temp. E31-N604A&E (B&F) Channel Functional," on the Division 2 Main Steam Line Tunnel Leak Detection System [SJ]. The leak detection system bypass switches [HS] for the Division 2 Reactor Core Isolation Cooling System (RCIC) [BN] and Reactor Water Cleanup System (RT) [CE] were in the bypass position as required by the procedure to support the surveillance test.

At about 1329 hours, while performing testing in accordance with CPS 9532.07, a Group 1 Containment Isolation occurred. The Group 1 Isolation caused the Main Steam [SB] Isolation Valves (MSIVs) [ISV] and Main Steam Line Drain Valves to automatically close, resulting in an automatic reactor scram.

Following the reactor scram, the Main Control Room (MCR) crew entered Emergency Operating Procedure (EOP)-1, "RPV Control," and off-normal procedure CPS 4100.01, "Reactor Scram." The Control Room Supervisor (CRS) directed operators to control reactor water level between level 3 and level 8 and reactor pressure between 800 and 1065 pounds per square inch gage (psig) using safety relief valves (SRVs) [RV]. The CRS is the Senior Reactor Operator that has command authority in the MCR. Because the MSIVs closed, the main condenser [COND] (the normal heat sink following a reactor scram) was not available for pressure control. Initially, following the scram, reactor pressure and level were maintained by automatic and manual operation of the SRVs; manual initiation of the RCIC pump [P] injecting into the vessel from the RCIC storage tank [TK] and operating RCIC in the tank-to-tank mode; Control Rod Drive System [AD] injection; and reactor feed supplied through the motor-driven reactor feed pump and condensate booster pumps.

At approximately 1542 hours, the MCR crew commenced activities to reestablish the main condenser as the heat sink in accordance with procedure CPS 3101.01, "Main Steam (MS, IS, ADS)," for reactor pressure control. The CRS gave a crew briefing outlining the plans for reestablishing the main condenser as a heat sink. No specifics were covered in the briefing regarding the task of warming the steam lines and opening the MSIVs. During performance of CPS 3101.01 section 8.1.1.3, "MSL Warmup/Unisolation when RPV is Pressurized MODE 2/3," the operator realized that the procedure section had steps for opening the inboard MSIVs, but not for opening the outboard MSIVs. Procedure section 8.1.1.3 was written with the assumption that the outboard MSIVs were already open; an earlier section of the procedure opens these valves. Operators initiated At Risk Revision (ARR) 00-0660 to change the procedure section to allow opening of the outboard MSIVs.

While the procedure change process was being pursued, reactor pressure was slowly increasing, and reactor water level was slowly decreasing. When reactor pressure increased to the discharge pressure of the condensate booster pump that was feeding the vessel, feed flow to the reactor vessel stopped and water level decreased more rapidly. At Risk Revision 00-0660 was approved at about 1842 hours and the outboard MSIVs were opened; however, reactor water level dropped to low level 3, initiating a reactor scram at about 1845 hours. At the time of this scram, the plant was in Mode 3 (HOT SHUTDOWN), reactor coolant temperature was approximately 500 degrees Fahrenheit, and pressure was approximately 700 psig.

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Following opening of the MSIVs, reactor pressure control was established by bleeding steam to the main condenser through the turbine bypass valves, and reactor water level control was established using the Condensate/Condensate Booster Systems [SD].

Investigation of this event identified that the Group 1 Isolation was caused by performance of the Division 2 Main Steam Line tunnel leak detection system surveillance at the same time that an unidentified failure of a Digital Signal Conditioning (DSC) card [IMOD] existed in the circuitry for the Division 1 Main Steam line tunnel ambient temperature channel. This condition satisfied the 2-out-of-4 logic for the Group 1 isolation. Maintenance work document AR F20376 was initiated to investigate and correct this issue. The investigation performed under work document AR F20376 identified that an input resistor on the DSC card had failed open. The investigation was unable to identify a conclusive cause for the resistor failure.

During the review of the scram response and recovery efforts the following human performance deficiencies/weaknesses were identified:

- Following the scram, the "B" MCR operator placed the Division 2 Leak Detection System bypass switches back to the normal position without first notifying the Control Room Supervisor. Placing the bypass switch back in the normal position while a main steam line temperature trip was still locked-in from performance of surveillance CPS 9532.07, resulted in a Division 2, Group 4 Reactor Water Cleanup System containment isolation that tripped the running RT "A" and "C" pumps.

- The "A" MCR operator did not manually trip the turbine driven reactor feed pumps (TDRFPs) prior to receiving the automatic high reactor water level 8 trip of the TDRFPs during the initial level transient following the scram as stated in off-normal procedure CPS 4100.01.

- While using the motor-driven reactor feed pump (MDRFP), the reactor vessel was overfed when feedwater regulating valve 1FW004 failed to respond to level control signals. This resulted in the MDRFP tripping on high reactor water level 8. The failure of 1FW004 to respond was caused by the "A" MCR operator inappropriately using the startup level controller for 1FW004 in manual with the push-buttons in the "double-detent" mode.

- During the scram recovery, coordination between operators in controlling reactor water level and pressure was poor and did not minimize the number of low reactor water level 3 trips that occurred during the initial level transients following the scram.

During the event, a relay [RLY] failure in the Reactor Recirc (RR) [AD] flow control valve [FCV] circuit caused a premature runback of the FCVs when reactor water level decreased to level 4. The relay issue was known and documented prior to the trip in Maintenance work documents AR F20511 and F24375. Repair of the faulty relay condition had been included in the scope for a forced outage due to the plant conditions required for the repair. The faulty relay condition was corrected prior to plant startup following this event.

Condition Reports 2-00-12-107 and 2-00-12-109 were initiated to track a cause and corrective action determination for the Group 1 Containment isolation, reactor scrams, and the operator human performance issues associated with this event.

TEXT (*If more space is required, use additional copies of NRC Form 366A*) (17)

No other automatic or manually initiated safety system responses were necessary to place the plant in a safe and stable condition. Other inoperable equipment or components did not directly affect this event.

CAUSE OF EVENT

The cause of this event is attributed to a lack of adequate indication or annunciation of main steam line high temperature and differential temperature trips. There is no indication or annunciation available for the technician to detect a pre-existing fault in a channel circuit.

A contributing cause of the Group 1 containment isolation and subsequent reactor scram was an inadequate procedure. Surveillance procedure CPS 9532.07 had inadequate provisions for preventing a pre-existing fault in an alternate channel from completing the Group 1 actuation logic. The procedure did not require use of the sensor bypass switch for the channel in test. Bypassing the channel in test would have prevented completion of the actuation logic for the Group 1 containment isolation.

The primary cause for the reactor scram on low water level while attempting to establish reactor pressure control by bleeding steam to the main condenser was poor command and control of this evolution by the Control Room Supervisor. This deficiency resulted in a loss of reactor water level and pressure control while a procedure revision was being pursued to allow opening the MSIVs to establish the main condenser as a heat sink. The loss of level and pressure control resulted in a reactor scram on low water level 3.

An additional cause for the reactor scram while attempting to establish reactor pressure control via the main condenser was an inadequate procedure. System operating procedure CPS 3101.01 sections for main steam line startup were formatted for specific sections to be performed independently for the plant condition. The section initially used by the operator during this event to open the MSIVs, Section 8.1.1.3, "MSL Warmup/Unisolation When RPV is Pressurized MODE 2/3," had provisions for opening only the inboard valves and assumed that the outboard MSIVs were already opened by a previous procedure section. This procedural inadequacy substantially delayed efforts in restoring the main condenser as a heat sink to control reactor pressure, resulting in reactor water level decreasing to the low water level 3 scram setpoint.

Contributing causes for the reactor scram on low water level include an inadequate pre-job briefing, a failure to adequately review the procedure prior to use, and inadequate control room crew teamwork. An inadequate pre-job briefing and the failure to adequately review the procedure prior to initiating actions to establish the main condenser as a heat sink resulted in the failure to identify the procedure deficiency prior to starting the evolution. Crew teamwork was inadequate because two control room operators had concerns about level and pressure control but did not question the CRS when pressure went out of the band high. The operators did not bring up the concern that level would become a problem as reactor pressure rose to the shutoff head of the condensate booster pump.

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CORRECTIVE ACTIONS

The DSC card that failed was replaced in accordance with maintenance work document AR F20376.

A design change was developed and installed during refueling outage 8 in the main steam line high temperature and differential temperature circuits to include indication of channel trips.

I&C Maintenance reviewed Reactor Protection System and Nuclear Steam Supply Shutoff System logic circuits to determine the plant's exposure to similar pre-existing circuit faults that can result in a reactor scram or Group 1 Containment isolation during surveillance testing. The review identified nine surveillance procedures having the potential to be affected by similar faults that can cause a reactor scram or Group 1 Containment isolation. The nine procedures have been revised to include appropriate precautions to prevent similar events.

System operating procedure CPS 3101.01 has been revised to improve human factors and to provide clear instructions for opening both the inboard and outboard MSIVs.

During the events described in this LER and follow-up critique, several operator and control room supervisor human performance issues were identified in areas such as command and control, control of resources, communications, equipment operation, and pre-evolution planning. In response to these issues, the affected members of the Main Control Room crew during these events were provided remedial training in the simulator with acceptable results. In addition, the Main Control Room crew was counseled on these issues.

The Director of Operations met with several Operations shift managers and control room supervisors at an offsite location to review Operations performance as documented by the NRC, Exelon Regional Operating Group, and site organizations. During this meeting, a comprehensive corrective action plan was developed to address the operator performance issues.

All Operations shift crews were trained on this event during the first requalification cycle of 2001. The training included a discussion of the events of the scram, the inappropriate actions, the positive behaviors, equipment deficiencies noted, the skill and rule based errors, the causal factors, and recommended corrective actions.

Teamwork training was provided to Operations crews.

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ANALYSIS OF EVENT

This event is reportable under the provisions of 10CFR50.73(a)(2)(iv) due to the actuations of the Reactor Protection System [JC] and engineered safety features.

This event was compared with Updated Safety Analysis Report (USAR) Section 15.2.4, "MSLIV Closures," and the Transient Safety Analysis Design Report, and was found to be less severe than the analyses.

ADDITIONAL INFORMATION

The Digital Signal Conditioner circuit card discussed during this event is model 147D8461G010, manufactured by General Electric Company.

A relay failure in relay logic card 1B33K649A4 caused the premature runback of the reactor recirculation flow control valves. The relay card is model 2AO+L2C-R, manufactured by the Foxboro Company.

A review of previously reported events did not identify any recent similar events.

No safety system functional failures occurred during this event.

For further information regarding this event, contact J. C. Wemlinger, Corrective Action Coordinator Lead, at (217) 937-3846.